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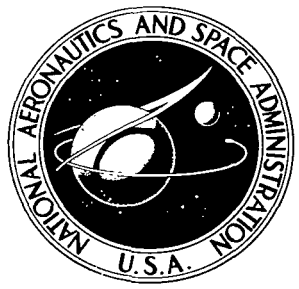

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CAUSES, MEASUREMENT, AND ELIMINATION OF FUEL-ELEMENT FISSION PRODUCT LEAKAGE AT THE NASA PLUM BROOK REACTOR

by A. Bert Davis and Richard H. Brickley

Lewis Research Center
Cleveland, Ohio

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ABSTRACT

Fission product leakage from the fuel elements of the Plum Brook Reactor (PBR) Facility resulted from corrosion and physical damage of the elements. The causes of the corrosion and physical damage are discussed. A method of measuring fission product release from irradiated elements and a method of remote disassembly and examination of leaking elements in the PBR Hot Laboratory Facility are described. The use of the iodine-133 to iodine-131 (I^{133}/I^{131}) ratio and a nitrogen-17 (N^{17}) fission product monitor in detecting the injection and source of fission products in the reactor cooling water system are discussed.

CAUSES, MEASUREMENT, AND ELIMINATION OF FUEL-ELEMENT FISSION PRODUCT LEAKAGE AT THE NASA PLUM BROOK REACTOR

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SUMMARY

Fission product leakage to the Plum Brook Reactor Primary Cooling Water (PCW) System occurred in five consecutive reactor cycles from June 1 to September 23, 1968. This report discusses the source and causes of this leakage and the methods used to determine the source and causes.

The source of the fission products was leaking fuel elements. The leakage was caused by (1) physical damage resulting from foreign particles in the PCW System and (2) pitting-type corrosion of the fuel elements. The foreign particles resulted from major core beryllium replacement. An electrical discharge machine had been used to remove reactor core beryllium structures which had failed. The resulting particles were satisfactorily removed by a 1/32-inch- (0.795-mm-) hole-size filter. The pitting-type corrosion resulted from extended storage of elements (between uses in the reactor) in water. Elements stored for 70 days in water of 5 to 30 micromhos per centimeter, 10 days in water of 3 to 5 micromhos per centimeter, and 110 days in water of less than 3 micromhos per centimeter had cladding failure. Elements stored for 225 days in water of less than 3 micromhos per centimeter also had cladding failure, but not as extensive.

A dry sipping apparatus was successfully used to locate leaking irradiated elements. The elements were placed in an air-filled chamber and allowed to heat to 300° F (420 K). A sample of the air was taken and analyzed for xenon-133 (Xe^{133}).

The iodine-133 to iodine-131 (I^{133}/I^{131}) ratio and data from a nitrogen-17 (N^{17}) fission product monitor were valuable in detecting fission product injection into the PCW and in determining whether the source was fuel-element leakage or system uranium contamination. The application of the I^{133}/I^{131} ratio and the N^{17} fission product monitor are described in this report.

Two fuel elements were remotely disassembled in the PBR Hot Laboratory Facility. The apparatus used for this disassembly and the results of the examination performed are discussed. Photographs of the pitted and damaged elements are presented.

INTRODUCTION

Fission product leakage from the Plum Brook Reactor (PBR) fuel elements was experienced in reactor cycles 76 through 80, during the time period June 1 to September 23, 1968. Since the magnitude of the leakage was small compared to that which has been experienced at other reactors, this aspect of the problem may not be of much interest. However, the causes of the leakage and the methods used to determine the source and the causes should be quite helpful to operators of other reactors.

The report is divided into four sections and three appendixes. The first section covers background information necessary to understand the remaining parts of the report. This is followed by sections covering (1) a summary of investigations made in reactor cycles 76 through 80 to determine the source of the fission products, (2) the analyses and investigations made to determine the cause of the problem after the source was identified, and (3) the conclusions reached as a result of the effort. Appendix A describes how the I^{133}/I^{131} ratio and the N^{17} fission product monitor were used in detecting the injection and source of fission products. Appendix B describes a dry sipping apparatus used to locate leaking irradiated fuel elements. Appendix C describes a disassembly apparatus used to remotely disassemble two fuel elements in the PBR Hot Laboratory Facility.

The most important features of the report are the determination of the causes of the fission product leakage and the methods used in determining the source and causes. One cause was physical damage of the fuel elements. The method used to eliminate the damage mechanism is covered. The second cause was pitting-type corrosion of the elements. The environment and the time in this environment which resulted in the corrosion are covered; however, the reasons for the corrosion are not explored in depth.

The method by which the N^{17} fission product monitor data and the I^{133}/I^{131} ratio were used to assess the source of fission product release to the PCW System is discussed. The interpretation of the data is related to reactor cycle operating experience.

Detailed information on the design and operation of a dry sipping apparatus is presented. The results obtained in dry sipping irradiated fuel elements are tabulated.

The apparatus used to remotely disassemble two leaking fuel elements is discussed in detail. Photographs of the disassembled elements are presented along with the results of the examination.

BACKGROUND INFORMATION

Facility Description

The PBR is a 60-megawatt-thermal pressurized-water test reactor. The reactor core is mounted in a tank 9 feet (2.74 m) in diameter by 30 feet (9.14 m) high (fig. 1).

The core components are shown in figure 2. The fueled section of the core consists of a 3 by 9 array of 22 MTR-type standard fuel elements and 5 control rods with fueled follower sections.

The PBR Primary Cooling Water (PCW) System has a volume of about 30 000 gallons (113.4 m^3) and a flow rate of about 17 800 gallons per minute (1.12 m^3/sec). It contains a bypass cleanup system and a full flow strainer. The bypass cleanup system contains two mixed bed ion exchangers. The flow rate through the system is about 100 gallons per minute (6.3×10^{-3} m^3/sec). The full flow strainer is a basket-type strainer with 1/16-inch (1.59-mm) openings.

The standard fuel elements are assemblies of 18 curved aluminum-clad plates of an aluminum-enriched uranium alloy. The fuel-plate spacing in an element is 0.115 inch (2.93 mm). The spacing between plates of adjacent elements, when they are installed in the core, is 0.120 inch (3.06 mm). The fuel-plate cores are 0.020 inch (0.51 mm) thick and consist of highly enriched uranium alloyed with aluminum alloy 1060, ASTM B-209-62. The 0.020-inch (0.51-mm) aluminum cladding is alloy 1100, ASTM B-209-62. The control rod fueled sections contain 14 curved aluminum plates which are identical to the standard fuel-element plates except for length. When new, the standard fuel elements contain 240 grams of uranium-235 (U^{235}), and the control rod elements contain 186 grams of U^{235}. The elements have been supplied by three different vendors since the initial operation of the PBR. Fuel burnup is presently limited to 113 megawatt days (9×10^{20} fissions/cm^3) per element. The maximum calculated heat flux to the PCW System is 1.72×10^6 Btu per hour per square foot (5.45×10^6 J/m^2/sec). The maximum calculated fuel-element temperature at this heat flux is 331° F (440 K).

The fuel elements are used in at least three reactor cycles before becoming depleted to the established operating limit. These can be three consecutive cycles or three or more nonconsecutive cycles depending on planned or unplanned deviations from the PBR standard-length cycle. Between nonconsecutive uses the elements are stored in containers in a water-filled canal. The storage containers are constructed of stainless steel and cadmium, both of which are exposed to the canal water.

Beryllium Replacement

When beryllium is exposed to fast neutrons, embrittlement and gas formation result. This gas formation caused the south beryllium plate (fig. 2) of the PBR core box to bow toward the core. After about 37 000 megawatt days (3.2×10^{12} kW-sec) of reactor operation, the embrittled plate cracked. Since the beryllium plate was installed with nonaccessible bolts and because it had swelled, the plate was removed by a remote electrical discharge machining (EDM) method. More information on the beryllium distortion and replacement can be found in reference 1. A chip collection system and an underwater vacuum cleaning system were used to keep the PCW System clean during the EDM operations.

Limiting Fission Product Levels and Methods of Measurement

The activity level of the PCW is limited to 1×10^{-4} microcurie per milliliter (3.7×10^{6} dis/sec/m^3) of strontium-90 (Sr^{90}) and 1×10^{-2} microcurie per milliliter (3.7×10^{8} dis/sec/m^3) of I^{131} by the Technical Specifications for the PBR. We routinely determine the activity levels of I^{134} and Sr^{92} and use an established relation between them and I^{131} and Sr^{90} to stay within the Technical Specifications. The relatively short half-lives of I^{134} and Sr^{92} permit a quicker assessment of fission product leakage. We also confirm the levels of I^{131} and Sr^{90} by radiochemistry analysis of a PCW sample taken immediately after reactor shutdown.

Two methods which proved to be very valuable in assessing fission product injection into the PCW are discussed in appendix A. These are an N^{17} fission product monitor and the I^{133}/I^{131} ratio. The interpretation and application of data obtained by these methods to the solution of the fission product leakage problem at PBR should be quite helpful.

Wet Sipping Method

A wet sipping method has been used at PBR to locate leaking fuel elements. The reactor is operated at 50 kilowatts. A probe, through which 100 milliliters per minute of water is drawn, is inserted above the fuel-element channels. The water is passed through an anion bed which is subsequently analyzed for fission products.

This method gave erratic results and was not successful. This was probably attributable to the low reactor power and the small fuel-element defects. These factors made it impossible to distinguish between fission products from PCW contamination and a true fuel-element leak.

Dry Sipping Method

A dry sipping method (refs. 2 and 3) has been successfully used at PBR to locate leaking fuel elements after they are removed from the reactor. Fuel elements are permitted to heat in an air-filled chamber to 300° F (420 K) at 1-atmosphere pressure. A gas sample is taken and analyzed for Xe^{133}. The results are reported in microcuries per milliliter per minute (dis/sec/m^3) of Xe^{133} at time of reactor shutdown. A detailed description of the apparatus is presented in appendix B.

DETERMINATION OF FISSION PRODUCT SOURCE

The fission product levels in the PCW were above normal in reactor cycles 76 through 80. Cycle 76 was the first cycle after the reactor beryllium sideplates were replaced.

The I^{134} level for a typical cycle with no leaking fuel elements varies in the range of 1000 to 3500 disintegrations per minute per milliliter (1.67×10^7 to 5.85×10^7 dis/sec/m^3). The I^{134} level for cycle 80 is shown in figure 3. Cycle 80 had higher fission product levels (140 000 dis/min/ml (2.34×10^9 dis/sec/m^3)) than any other cycle in the history of PBR operation. As is obvious from the magnitude of the I^{134} levels in figure 3, the levels were kept within allowable, safe limits by reactor power decreases and cycle terminations.

The primary goal, beginning with cycle 76, was to choose the course of action which would most quickly identify the source or sources of the fission product leakage. This was done by using the following methods to gather information:

(1) Operating the reactor with various fuel-element and fueled experiment loadings and observing I^{134} levels, I^{133}/I^{131} ratios, and the N^{17} fission product monitor

(2) Using the dry sipping apparatus in a standard controlled way to determine the Xe^{133} release from irradiated elements (This apparatus was built and first used between cycles 77 and 78.)

(3) Visually examining irradiated elements with binoculars through about 10 feet (3 m) of water shielding

The possible causes of the fission product leakage were believed to be

(1) A random fuel-element leak (Previous experience at PBR showed that minor fission products resulted from this cause about once every eight reactor cycles.)

(2) An unidentified cause associated with the beryllium plate replacement

(3) Reuse of a leaking element from a cycle prior to cycle 76 (This prior cycle would have had higher than normal I^{134} levels, yet not as high as those in cycles 76 through 80. The leaking element could be deteriorating with added use.)

(4) Defective elements from a new fuel-element vendor (This could not be the sole cause since elements from vendor 3 were not used in the first cycle-76 loading, nor were they used prior to cycle 76.)

(5) Failure of a fueled experiment

A summary of the results obtained and conclusions reached in cycles 76 through 80 is discussed in the following paragraphs. This information may be useful to the reader in developing and executing a plan should he be faced with a fission product release to a reactor cooling water system.

From cycle-76 operation with three fuel-element loadings, a wet sipping of a cycle-76 core, and visual examination of the core and elements removed from the core, the following conclusions were reached:

(1) The wet sipping method was not effective in locating leaking fuel elements.

(2) Damage was being done to the elements by foreign particles in the PCW. In spite of the efforts to maintain system cleanliness described in the section Beryllium Replacement, these foreign particles were most likely beryllium chips left in the system after the beryllium plate replacement. The damage was attributable to foreign particles because a path of corresponding gouges was found on exterior plates of adjacent fuel elements. It was apparent that a particle had worked its way down between these elements, causing damage to both.

As a result of these findings, the wet sipping operation was discontinued and a full core straining operation was performed. This straining consisted of substituting a series of screens with hole size varying from 1/4 to 1/32 inch (6.35 to 0.795 mm) for the 3-by-9 array of fuel elements. The PCW System was then operated until no further chips were collected on the screens. This required 11 hours.

Cycle 77 also contained a fission product source. From cycle-77 operation, either of the following conclusions was possible:

(1) There was a cause of fission product leakage in addition to the beryllium damage done in cycle 76, one of causes 1, 3, 4, or 5 previously described.

(2) A visual examination of elements used in cycle 76 was not a satisfactory way to separate damaged and undamaged elements. Cycle 77 contained some visually acceptable elements from cycle 76.

Prior to cycle 78, the dry sipping apparatus was used to select the cycle fuel loading. In spite of this, cycle 78 contained a fission product source. From the cycle-78 operation, either of the following conclusions was possible:

(1) Causes 1, 3, 4, or 5 were causing fission products to be released to the PCW.

(2) The dry sipping apparatus was not permitting separation of damaged from undamaged elements.

The second conclusion was believed to be more probable because the apparatus was new, the operating procedures were not properly followed, and there was little experience in operating the apparatus.

During cycles 79 and 80, various fuel loadings were used, and irradiated elements were dry sipped and visually examined in a controlled, standard manner. The following conclusions were reached:

(1) Cause 3 was eliminated. Core loadings with no elements from pre-76 cycles, which had higher than normal I^{134} levels, still contained a source of fission products.

(2) Cause 4 was eliminated. A core loading containing no elements from vendor 3 still contained a source of fission products.

(3) Cause 5 was eliminated. Experiment performance made this an unlikely contributor. Operation with the experiments removed still contained a source of fission products.

(4) Cause 1 was unlikely because the frequency of the random failures would have had to increase markedly.

(5) The dry sipping method was an effective way to locate leaking fuel elements.

Table I shows the results of elements which were dry sipped. Elements which are considered to be definite leakers are B-130, B-135, B-146, B-203, and BC-15. Although all elements used in cycles 76 through 80 were not dry sipped, at least one of these five elements was in each cycle.

At the conclusion of cycle 80, all of the five postulated causes had been eliminated. However, examination of the history of the elements found to be leakers in the dry sipping operation showed element age to be a common factor for all elements except B-130. Element B-130 was considered to have been damaged in cycle 76. The other elements were not in cycle 76. Therefore, the age of an element (i.e., the time since its first use in the reactor) was determined to be a factor in fission product release to the PCW.

One concern during cycles 76 through 80 was that a sudden burst of fission product release to the PCW might occur. This was not experienced. Although an increasing injection rate to the PCW was experienced, there was always time (about 1 to 2 hr minimum) to react to this increase and maintain acceptable fission product levels.

Beginning with cycle 81, fission product injection into the PCW from leaking fuel elements was eliminated. This was done by avoiding elements which were used between the time of beryllium replacement and the full core filtration and avoiding elements which were old in terms of time since their first use.

One point of interest during the period of fission product leakage was the system contamination level being established. Cycle 81, which contained no leaking fuel element, provided this information. The I^{134} level, as a result of system uranium contamination, decreased from about 30 000 to 20 000 disintegrations per minute per milliliter (5×10^8 to 3.34×10^8 dis/sec/m^3) during the cycle. Two cycles later, the level had dropped to about 10 000 disintegrations per minute per milliliter (1.67×10^8 dis/sec/m^3).

DETERMINATION OF CAUSE OF FISSION PRODUCT RELEASE FROM RELATIVELY OLD ELEMENTS

The following investigations were begun to identify the cause of leakage from the old elements:

(1) Evaluations were made to determine if correlations existed between Xe^{133} release from these relatively old elements in the dry sipping apparatus and the following parameters: average and peak-to-average fuel-element burnup, element storage time in water and water chemistry during this storage time, and element fabrication date.

(2) A correlation of common characteristics or factors for all leaking elements was made.

(3) Elements B-130 and B-203 were disassembled in the PBR Hot Laboratory to determine the appearance of these leaking elements. (See appendix C for a description of the apparatus and method used to disassemble the elements.)

No correlations were found with average and peak-to-average burnup or element fabrication date. A correlation between element leakage and storage time in water was found. Figure 4 shows that the Xe^{133} release rate in the dry sipping operation is less than 10×10^{-5} microcuries per milliliter per minute (6.2×10^{4} dis/sec/m^3) for a water storage time of less than 190 days. This storage time does not include the time the elements were operated in the high-quality PCW System. Although the 10×10^{-5} microcuries per milliliter per minute (6.2×10^{4} dis/sec/m^3) cannot be stated as a definite dividing point between a leaking element and a contaminated element, it is close to the proper value. The water during this storage period varied between 1 and 30 micromhos per centimeter. It ranged between 5 and 30 micromhos per centimeter for 70 days and between 3 and 5 micromhos per centimeter for 10 days of this storage period. Otherwise it was below 3 micromhos per centimeter.

Another data point on the effect of water storage time on fuel-element fission product release was obtained in reactor cycle 87. To improve fuel-element utilization we purposely used elements that had been stored in water as long as 225 days. The water conductivity was at all times less than 3 micromhos per centimeter. The N^{17} fission product monitor and the I^{133}/I^{131} ratio both showed that this cycle contained leaking fuel elements. However, the leaks were not large; the I^{134} levels in the PCW did not exceed 35 000 disintegrations per minute per milliliter (5.84×10^{8} dis/sec/m^3). These elements were not dry sipped.

A correlation may also exist between element storage time in water and element contamination pickup during a reactor cycle as measured by the Xe^{133} release rate in the dry sipping operation. New elements were generally contaminated to levels of less than 5×10^{-5} microcurie per milliliter per minute (3.1×10^{4} dis/sec/m^3), while the older elements were generally contaminated to levels between 5×10^{-5} and 10×10^{-5} microcurie per milliliter per minute (3.1×10^{4} and 6.2×10^{4} dis/sec/m^3) when these elements were

operated in the same cycles. This higher value of the older elements, however, may also be the beginning phases of fission product leakage or a combination of leakage and increased contamination. No correlation with other common characteristics or factors was found.

Perhaps the most interesting findings came from the Hot Laboratory disassembly of elements B-130 and B-203. The individual plates were removed from each element (see appendix C). The plates were then examined remotely with a Kollmorgen periscope and a stereomicroscope.

Several plates from element B-130 are shown in figures 5(a) to (d). Figure 5(a) shows a typical gouge on the convex side of a plate. See page 6 for a discussion of gouges found on exterior plates of fuel elements. A 0.015-inch- (0.38-mm-) deep gouge was found in the fuel meat region. A 0.040-inch- (1.02-mm-) deep gouge was found adjacent to the fuel meat region. The fuel-cladding thickness is 0.020 inch (0.51 mm). Figure 5(b) shows gouge paths on the convex side of a plate. There is evidence of corrosion beginning along the gouge paths. Figure 5(c) shows typical corrosion on the concave side of the plates of this element. Figure 5(d) shows a gouge path with corrosion along the path. The corrosion along the gouge paths apparently occurred during storage of the element after the gouges had occurred. The corrosion on the concave side of the plates is not associated with a location of mechanical damage.

The finding of gouges on the interior plates of element B-130 requires a revision to an analytical conclusion that enough force was not present to cause damage by foreign particles. Vibration of the plates against entrapped foreign particles may be the explanation.

Several plates from element B-203 are shown in figures 6(a) to (b). All of these plates show pitting-type corrosion. Figure 6(a) shows pits, on the concave side of different plates, which had penetrated into the fuel region. Figure 6(b) shows a pit, on the concave side of a plate, which had not yet penetrated the depth of the cladding. All of these plates from element B-203 had a brownish-gold film on the lower half of the plates. The cation elements of these films were identical to the cladding material and its trace elements.

The pitting-type corrosion on both elements B-130 and B-203 supports the correlation found between Xe^{133} release in the dry sipping operation and the storage time of the elements in the poor quality water. Element B-203 had a longer storage time than B-130.

The fact that corrosion was almost exclusively restricted to the concave side of the plates has not been explained. However, it has been learned that different dies are used in the plate bending operation. A rubber die contacts the convex side, and a steel die contacts the concave side of the plates. It may be that foreign particles are imparted to the concave side of the plate by the steel die, although the vendor discounts this possibility.

The presence of the brownish-gold film on element B-203 and its absence on element B-130 have not been explained. It may be due to the longer water storage time of B-203 or to a larger release of fission products during in-pile operation by this element.

The hot laboratory examination led to the conclusion that element B-203 was a larger fission product source than was element B-130. But the dry sipping test gave the opposite results. This discrepancy was probably due to the large pits in element B-203 having released a larger portion of the fission products to the PCW, leaving less to be released in the dry sipping test.

CONCLUSIONS

Fission product release from fuel elements to the Plum Brook Reactor Primary Cooling Water (PCW) System was experienced in five consecutive reactor cycles. The causes of this release have been determined, and restrictions have been adopted to avoid a recurrence. The following conclusions were drawn from the solution of this problem:

1. Pitting-type corrosion of the fuel elements occurs from extended storage of the elements in water. When the conductivity of the storage water, in micromhos per centimeter, ranged from 5 to 30 for 70 days, 3 to 5 for 10 days, and below 3 for 110 days, the fuel-element cladding was violated. In water whose conductivity was below 3 micromhos per centimeter for 225 days, the cladding was also violated, although not as severely. The pH in both cases varied between 5.5 and 7.2.

2. Damage to fuel-element cladding with associated fission product release can result from the presence of small foreign particles in the PCW System. Straining with a 1/16-inch- (1.59-mm-) hole-size strainer did not avoid the damage. Straining with a filter of 1/32-inch (0.795-mm) hole size eliminated the damage.

3. A dry sipping method, described in this report, was used successfully to locate irradiated fuel elements which have fission product leaks.

4. A wet sipping method, also described in this report, was not successful in locating fuel elements which have fission product leaks.

5. The iodine-133/iodine-131 ratio and data from a nitrogen-17 fission product monitor were valuable in detecting fission product injection into the PCW and in determining whether the source was a leaking element or uranium contamination.

6. The effect of the fission product release on reactor operating efficiency was minimized by using several data producing operations simultaneously. These included

operating the reactor at power levels conducive to data generation, dry sipping irradiated elements, and visually examining irradiated elements.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, May 7, 1969,
122-29-05-11-22.

APPENDIX A

METHODS USED TO ASSESS FISSION PRODUCT RELEASE TO A REACTOR PRIMARY COOLING WATER SYSTEM

The Plum Brook Reactor (PBR) relies predominantly on two methods for determining the fission product injection into the PCW System. These methods are radiochemistry analyses of samples taken from the system and a nitrogen-17 (N^{17}) fission product monitor.

The N^{17} system uses two neutron detectors spaced about two N^{17} half-lives apart in the PCW System. The ratio of signals from the two detectors is constant when N^{17} is the neutron contributor. Delayed neutrons from fresh fission products cause a drop in the ratio. The presence of N^{17}, which normally produces an undesired background for delayed neutron detection, is used to advantage to give a constant indication that the system is operating properly. This system responds rapidly to the delayed neutrons from fresh fission products. A normal ratio indicates no fission product injection into the PCW. A below-normal ratio indicates a constant injection rate of fission products. A decreasing ratio indicates an increasing fission product injection rate.

Although the system ratio can be calibrated in terms of steady-state levels of fission products of interest, this had not yet been done at the time the PCW fission product problem occurred. The system had just been installed. Consequently, it provided valuable qualitative, but little quantitative, information during the fission product problem.

Radiochemistry analyses provided activities of the various fission product elements. The ratio of two of these fission product activities (I^{133} to I^{131}) was quite valuable in distinguishing between fission products from uranium contamination and those from fuel-element leakage during cycles 76 through 80.

The equilibrium ratio inside a nonleaking fuel element is calculated to be 2.2, which is the ratio of the fission yields of I^{133} and I^{131}. The equilibrium ratio in the PCW System from uranium contamination is calculated to be in the range of 10 to 15. Since this ratio is affected by both the decay of each isotope and the cleanup of the isotopes by the PCW bypass cleanup system, its magnitude is not precisely determined. It is believed to be between 10 and 15, however. For purposes of clarification, this range of 10 to 15 is hereinafter referred to as the normal range. From these two calculated results, the following statements can be made:

(1) For uranium contamination and no fuel-element fission product leak, the ratio would start at a high value and decrease to the normal value.

(2) For a new leaking element which does not reach an equilibrium level of I^{131} during a cycle, the ratio would start at a high value and decrease to a level between 2 and the normal value during the cycle.

(3) For a relatively old leaking element which contained no I^{133} but still contained some I^{131}, the ratio would start low, increase to a maximum, and then decrease to a level between 2 and the normal value.

From these statements it can be seen how this information may be used to assess the source of fission products.

Typical results obtained at PBR are as follows:

(1) The ratio in a good cycle with no fission product leakage decreases from about 16 to 11. This corresponds to item (1) in the preceding paragraph.

(2) Cycles with high I^{134} levels have started with a ratio of 14 and decreased to a level of 4. This corresponds to item (2) in the preceding paragraph.

(3) Other cycles with high I^{134} levels have started with a ratio of 5, increased to 12, and decreased to 4. This corresponds to item (3) in the preceding paragraph.

APPENDIX B

DRY SIPPING APPARATUS

The dry sipping apparatus shown in figure 7 consists of a test chamber (fig. 8) mounted 13 feet (3.96 m) below water, a chamber evacuation system, a gas collection and disposal system, a chamber fill system, and an emergency cooling system.

The chamber evacuation system supplies 7 to 9 psig (4.85×10^4 to 6.23×10^4 N/m^2 gage) regulated air through a 0.75-inch (1.9-cm) line to the chamber head to force the water from the test chamber. A bubbler drain on the bottom of the chamber is used to indicate that evacuation is completed.

The gas collection and disposal system consists of a nitrogen-charged, 300-psig (2.07×10^6-N/m^2 gage) accumulator, a 0.75-inch (1.9-cm) vent line, a 5-cubic-inch (8.2×10^{-5}-m^3) cylindrical sample container, and a 0.5-inch (1.27-cm) line to the off-gas cleanup system. Samples of the gas are analyzed for Xe^{133} activity.

The chamber fill system consists of a 1.0-inch (2.54-cm) line supplying 80-psig (5.5×10^5-N/m^2 gage) deionized water through a manually operated valve to the bottom of the test chamber. The displaced air is disposed of through the gas collection and disposal system to the off-gas cleanup system.

The emergency cooling system consists of a 1.0-inch (2.54-cm) line supplying 80-psig (5.5×10^5-N/m^2 gage) deionized water through a solenoid-operated valve to the bottom of the chamber. The solenoid valve is controlled by the set point recorder to open when the fuel-element sideplate temperature reaches 350° F (449 K). The manual valve in the chamber fill system is used as a backup for emergency cooling.

The following is a description of how the system was used:

(1) The test chamber was evacuated and a background sample of air obtained. If the Xe^{133} activity in this background sample measures $> 1\times10^{-6}$ microcurie per milliliter (3.7×10^4 dis/sec/m^3), the system is flushed and purged until this value is reached.

(2) A fuel element with a calculated heat generation of $< 2.4\times10^3$ Btu per hour (7.03×10^2 J/sec) is inserted into the test chamber, and the head is installed and bolted down. A heat generation rate of $< 2.4\times10^3$ Btu per hour (7.03×10^2 J/sec) ensures that the temperature of the fuel does not exceed 400° F (478 K).

(3) The chamber is evacuated by opening valve C and then valve A (fig. 7) and supplying air at 7 to 9 psig (4.85×10^4 to 6.23×10^4 N/m^2 gage) to force the water out through valve C.

(4) When bubbles are observed coming from valve C, both valves A and C are closed.

(5) The fuel-element sideplate temperature is allowed to rise to 300° F (422 K); then valve B is opened to allow cooling water to enter the chamber.

(6) A sample of the gas trapped in the chamber is obtained by venting one-half of the accumulator through valve J, opening valve H until a water level is seen in the sight glass, and then closing valve H. The accumulator is pressurized by closing valve J, opening valve K until the accumulator gage reaches 300 psig (2.07×10^6 N/m^2 gage), and then closing valve K. Valve L is opened, allowing the gas to flow into the sample container. When the sample container pressure gage reached 285 psig (1.96×10^6 N/m^2 gage), valve L is closed. The gas remaining in the system is discharged to the off-gas cleanup system through valve G. The sample container is removed for analysis.

(7) The test chamber head is removed and the tested fuel element moved to the storage cart.

APPENDIX C

FUEL-ELEMENT DISASSEMBLY

Disassembly of the fuel elements was accomplished by cutting off sections of the edge of the sideplates (fig. 9) and removing each fueled plate in cell 1 of the Hot Laboratory. The X-rays of the plates were carefully examined for fuel location and cutting clearances prior to disassembly. The fuel element was clamped to the table of a modified commercial 10-inch (25.4-cm) table saw driven by a 1-horsepower (0.746-kW), 3450-rpm (57.5-rps) motor (fig. 10). The ratio of cutting blade to motor speed was 1 to 1. The element was fed into the saw by a modified 0.5-inch- (1.27-cm-) capacity variable speed drill motor at 1 inch per minute (0.0424 cm/sec).

A slice, 0.115 inch (0.292 cm) wide and 0.188 inch (0.477 cm) deep, was made down the entire edge of each sideplate. This freed the top fueled plate from the sideplates. Identical slices were made through the width of the sideplates, freeing fueled plates as the slice progressed. It took approximately 1 hour to free each plate.

The saw was mounted in a large flat pan, and a vacuum cleaner was used to remove all chips. Health safety personnel monitored all cutting operations.

Following the Hot Laboratory examination, results of which are contained in the main text, the elements were reassembled using aluminum bolts, nuts, and spacers and returned to the fuel storage facility.

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1. Fecych, William: Experiences with Beryllium Swelling and Replacement in the Plum Brook Reactor. NASA TM X-1794, 1969.

2. Peterson, Richard H.: Fuel Assembly Cladding Defect Determination Using the "Dry Sipping" Technique. Pacific Gas & Electric Co. (AEC Rep. DOCKET 50-133), Feb. 24, 1966. (Available from AEC Public Document Room.)

3. Blakely, J. P., Compiler: Fuel-Assembly Cladding-Defect Determinations Using the "Dry Sipping" Technique. Nucl. Safety, vol. 7, no. 4, Summer 1966, pp. 511-513.

TABLE I. - FUEL-ELEMENT TEST RESULTS

Fuel element	Storage time in water, days	Xenon-133 release rate[a]		Fuel element	Storage time in water, days	Xenon-133 release rate[a]	
		μCi/ml/m	dis/sec/m^3/sec			μCi/ml/m	dis/sec/m^3/sec
B-054	594	10.6×10^{-5}	6.6×10^{4}	B-C42	1	3.3×10^{-5}	2.0×10^{4}
B-060	594	52.3	3.2×10^{5}	B-C43	0	1.5	9.4×10^{3}
B-068	560	18.7	1.2	B-C44	10	2.6	1.6×10^{4}
B-071	560	8.1	5.0×10^{4}	B-C49	26	1.1	6.5×10^{3}
B-127	7	1.1	6.9×10^{3}	B-C50	6	7.1	4.4×10^{4}
B-130	132	2300.0	1.4×10^{7}	E-002	58	2.1	1.3
B-134	10	1.3	8.0×10^{3}	E-005	62	3.3	2.1
B-135	224	208.0	1.3×10^{6}	E-006	58	.1	5.1×10^{2}
B-136	224	26.0	1.6×10^{5}	E-008	62	0	1.2
B-138	136	9.2	5.7×10^{4}	E-009	57	.2	1.2×10^{3}
B-141	157	3.6	2.2	E-010	8	1.0	6.4
B-142	194	11.5	7.1	E-011	4	3.9	2.4×10^{4}
B-143	155	2.3	1.4	E-012	4	3.1	1.9
B-144	138	8.6	5.3	E-014	4	9.9	6.1
B-145	63	.4	2.4×10^{3}	E-017	58	0	2.6×10^{2}
B-146	202	179.0	1.1×10^{6}	E-021	0	.1	5.4
B-147	63	6.7	4.1×10^{4}	E-022	4	0	2.4
B-150	157	3.1	1.9×10^{3}	E-023	3	.1	7.2
B-160	191	1.6	9.7	E-025	3	.1	7.2
B-169	6	1.7	1.0×10^{4}	E-026	4	.1	8.3
B-186	235	.2	1.1×10^{3}	E-027	4	1.3	7.7×10^{3}
B-190	235	3.1	1.9×10^{4}	E-028	5	.6	3.4
B-203	301	309.0	1.9×10^{6}	E-029	4	.2	1.2
B-204	301	61.3	3.8×10^{5}	E-031	5	0	1.9×10^{2}
B-209	268	14.2	8.8×10^{4}	E-032	5	.1	3.2
B-210	268	61.6	3.8×10^{5}	E-034	5	2.4	1.5×10^{4}
B-C15	288	252.0	1.6×10^{6}	E-038	3	0	6.5×10^{1}
B-C16	16	6.9	4.3×10^{4}	E-039	3	.2	1.3×10^{2}
B-C20	37	6.3	3.9	E-052	4	0	7.7×10^{1}
B-C28	63	8.6	5.3	E-059	4	0	6.7

[a]Normalized to shutdown of cycle last used.

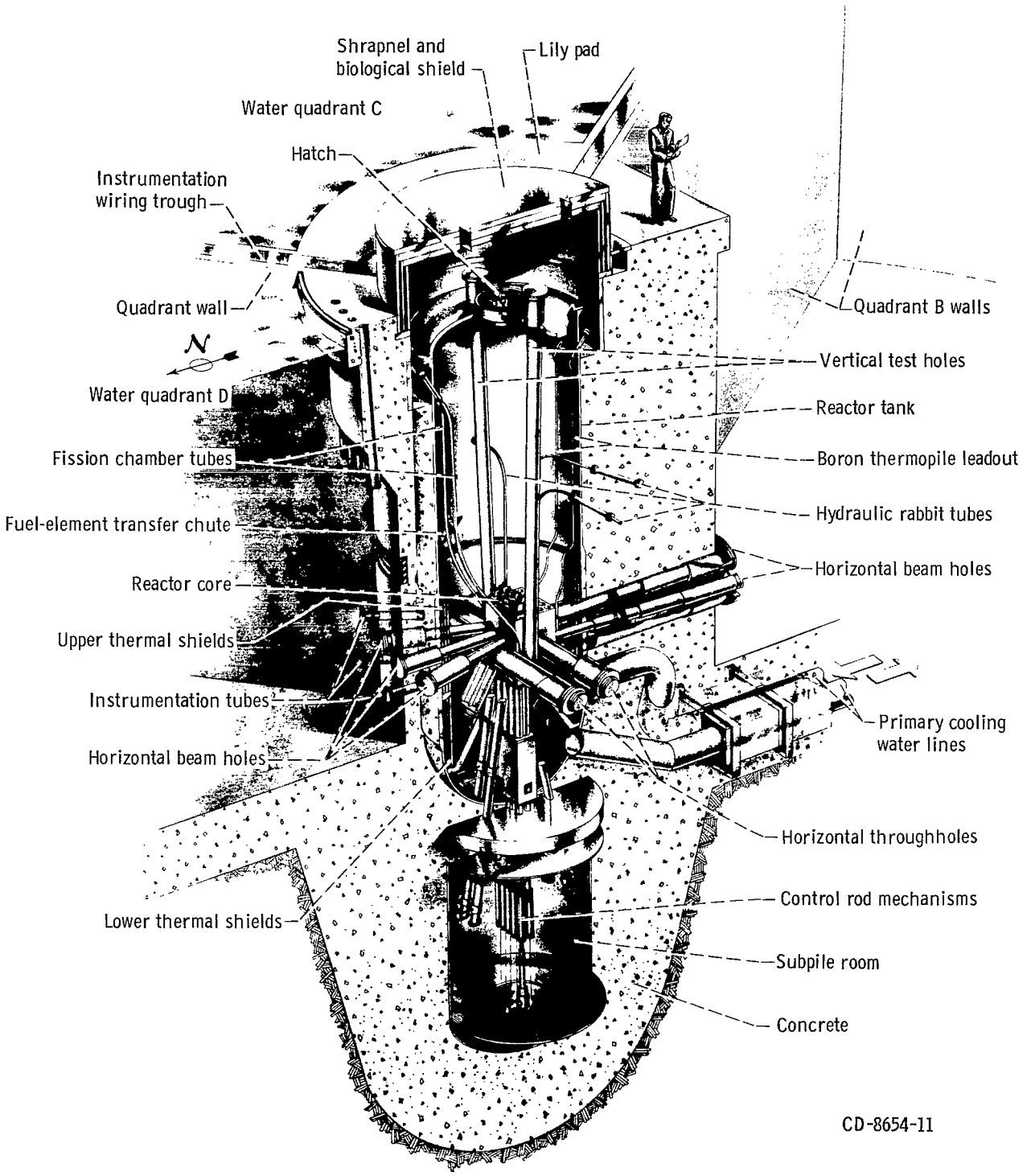
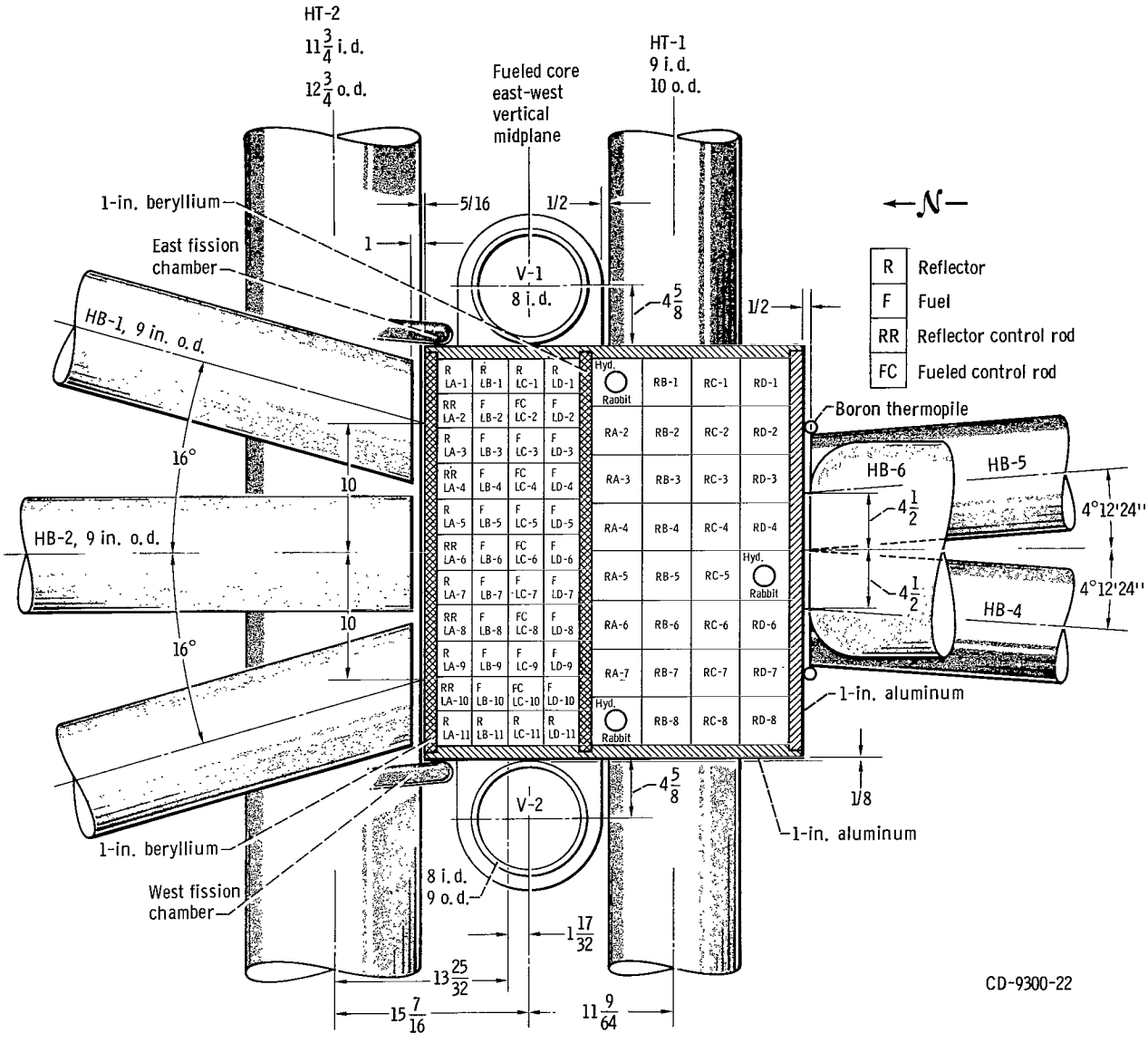
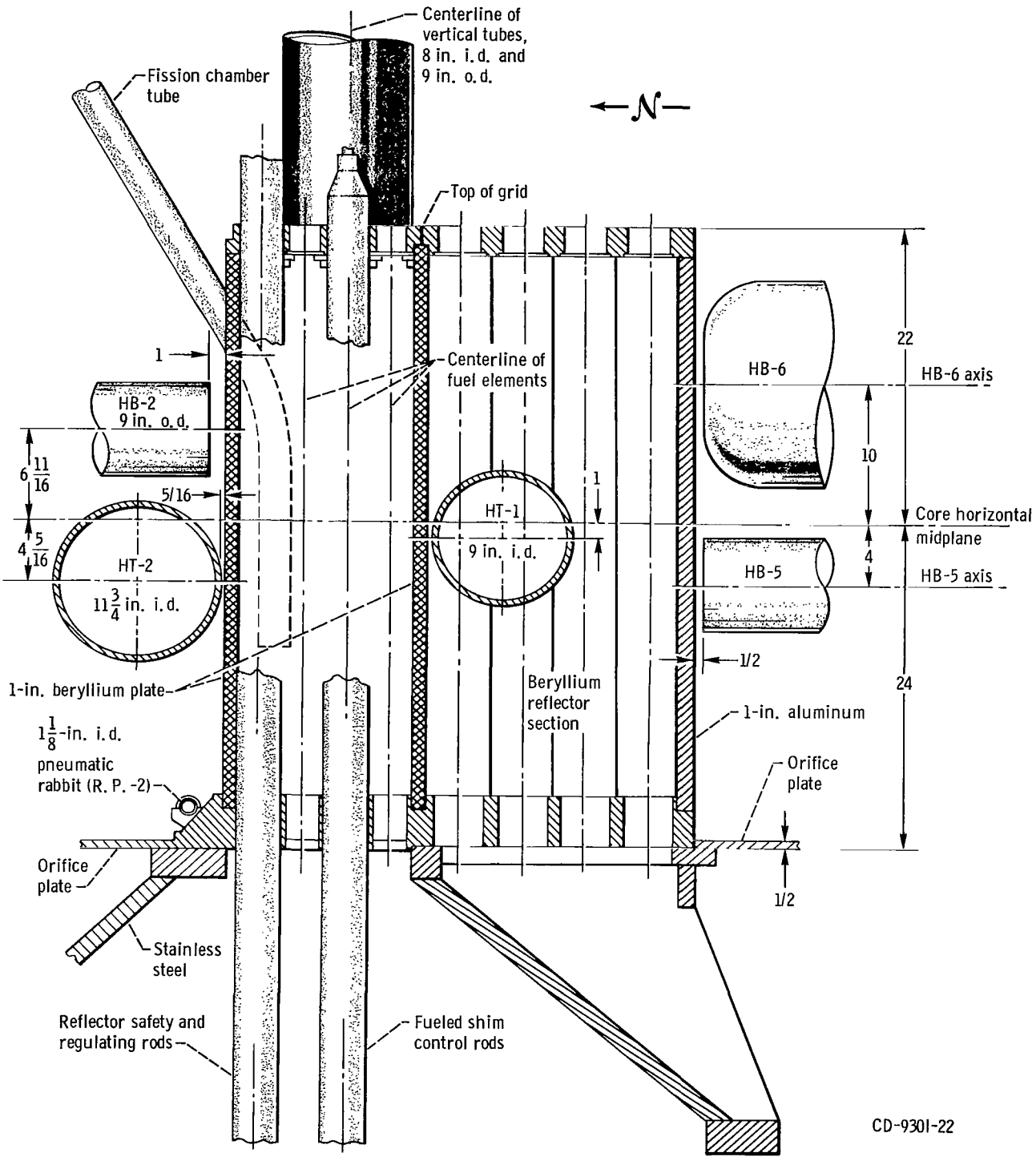


Figure 1. - Reactor tank assembly.



(a) Horizontal section.

Figure 2. - Reactor core. (Dimensions are in inches.)



(b) Vertical section.

Figure 2. - Concluded.

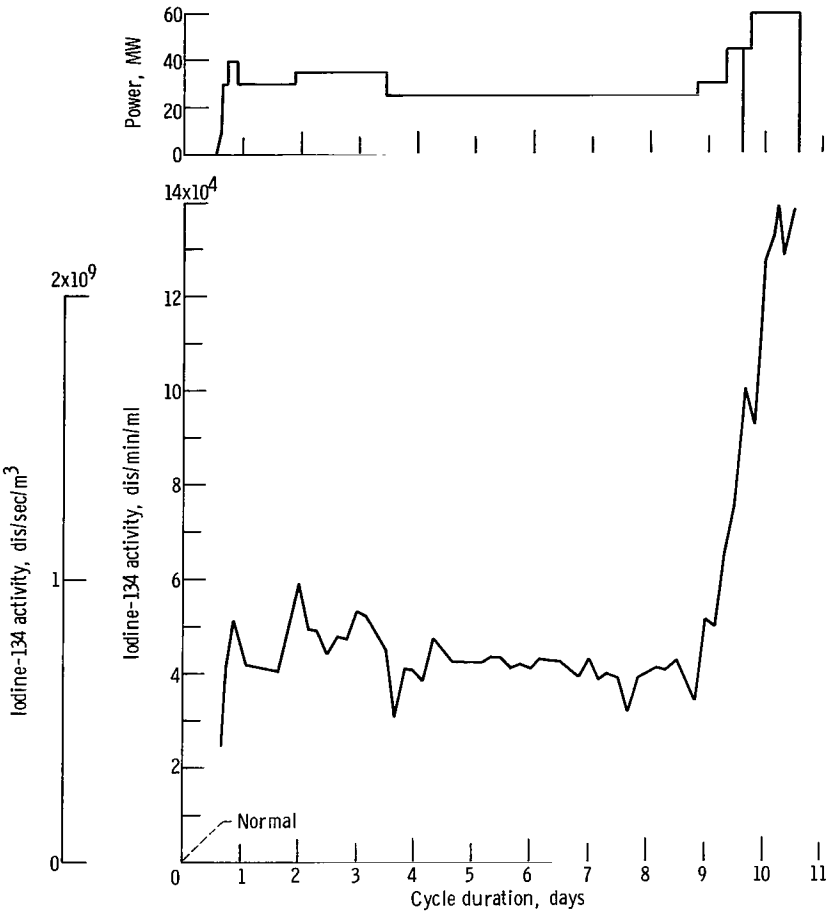


Figure 3. - Primary water activity. Cycle 80; core 3.

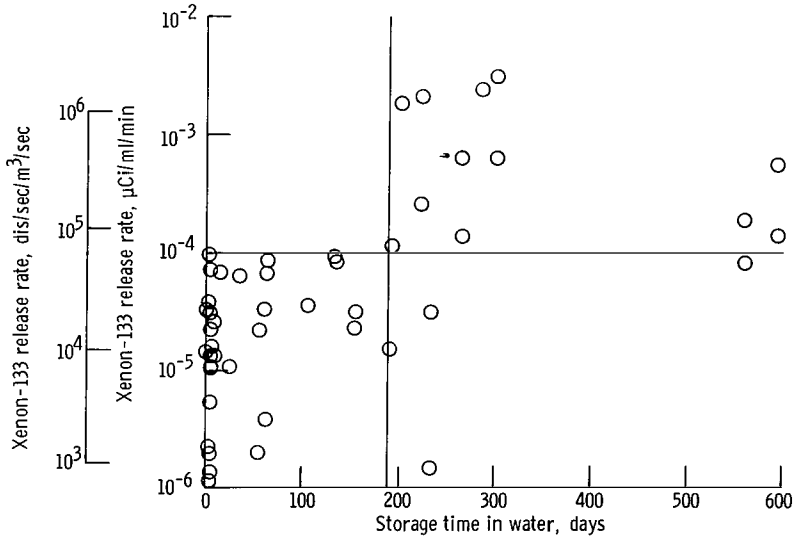


Figure 4. - Fuel-element test results.

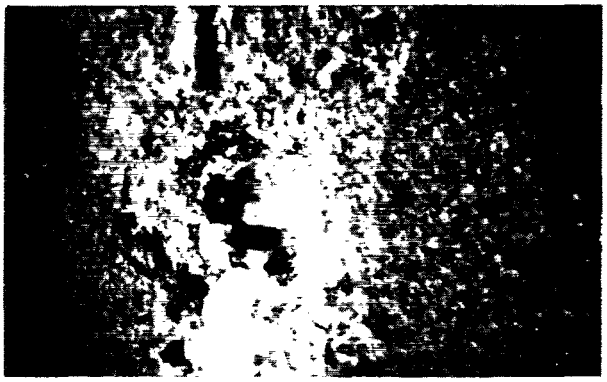
(a) Typical gouge, convex surface.

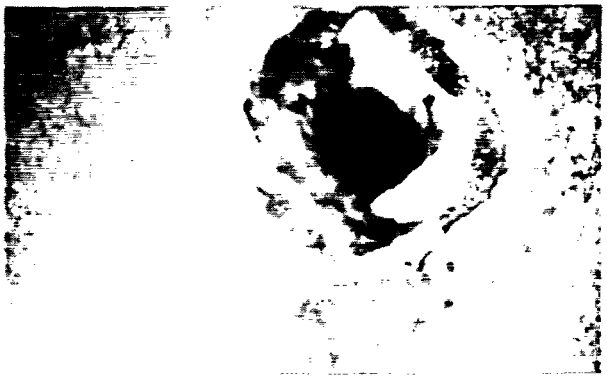
(b) Typical gouge path, convex surface.

(c) Typical corrosion, concave surface.

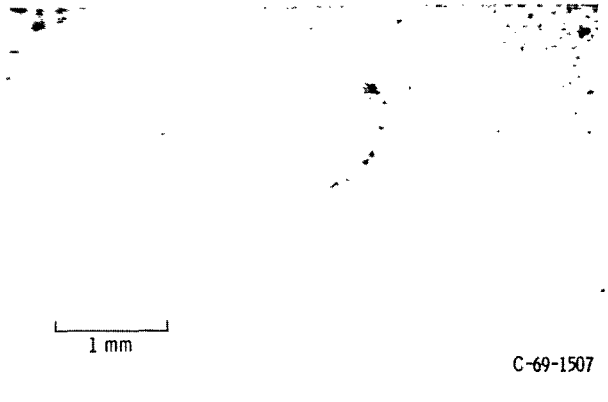
(d) Gouge path exhibiting initial corrosion, concave surface.

Figure 5. - Fuel element B-130, showing gouging and corrosion.



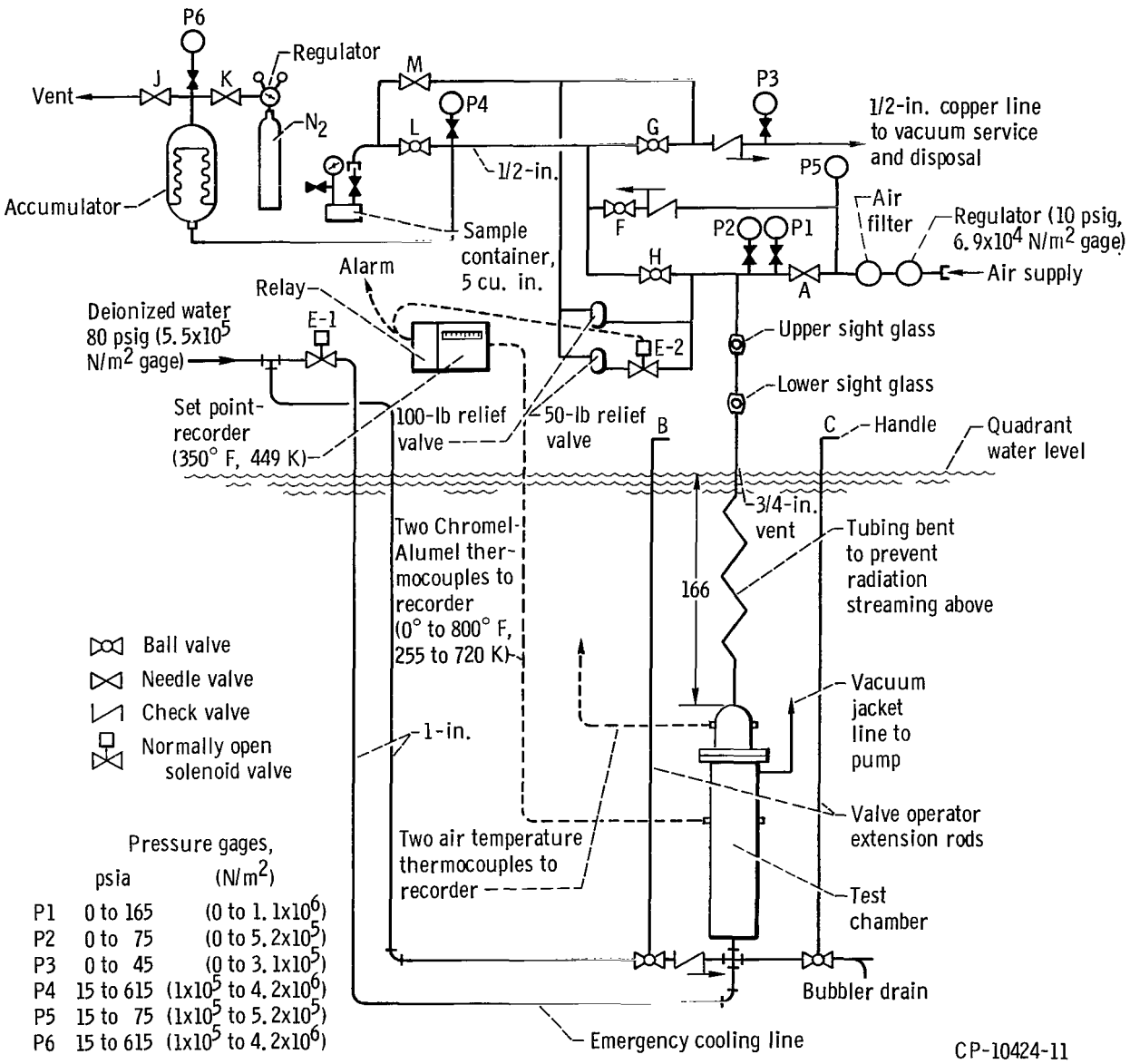


(a) Typical corrosion pits showing penetration, concave surfaces.



(b) Initial stage of corrosion.

Figure 6. - Fuel element B-203 showing corrosion.



Pressure gages,

	psia	(N/m^2)
P1	0 to 165	(0 to 1.1x10^6)
P2	0 to 75	(0 to 5.2x10^5)
P3	0 to 45	(0 to 3.1x10^5)
P4	15 to 615	(1x10^5 to 4.2x10^6)
P5	15 to 75	(1x10^5 to 5.2x10^5)
P6	15 to 615	(1x10^5 to 4.2x10^6)

Figure 7. - Dry sipping apparatus. Dimensions are in inches.

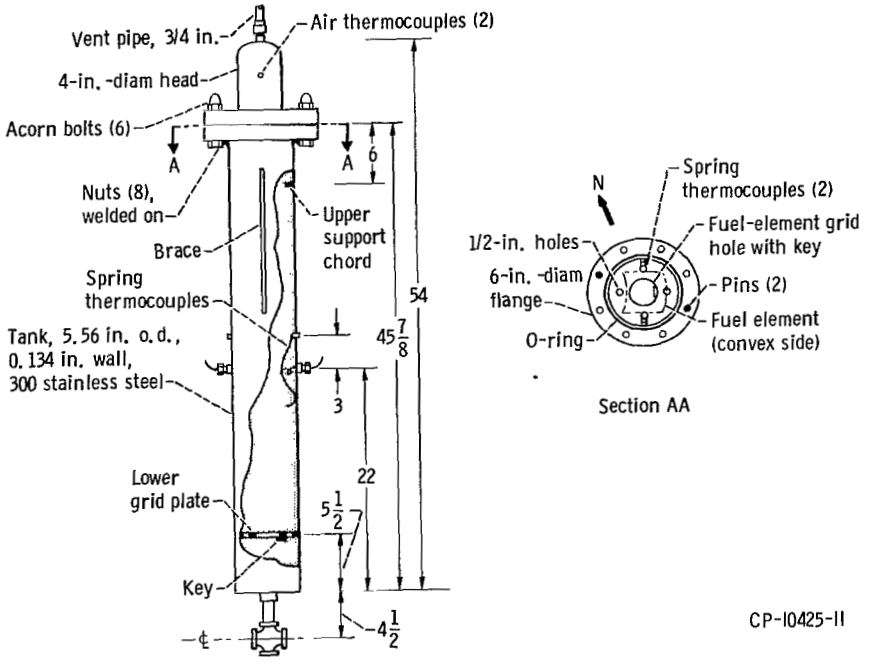


Figure 8. - Test chamber. Dimensions are in inches.

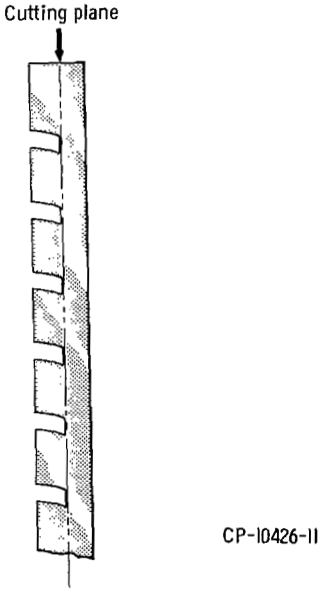


Figure 9. - Sideplate cutting plane.

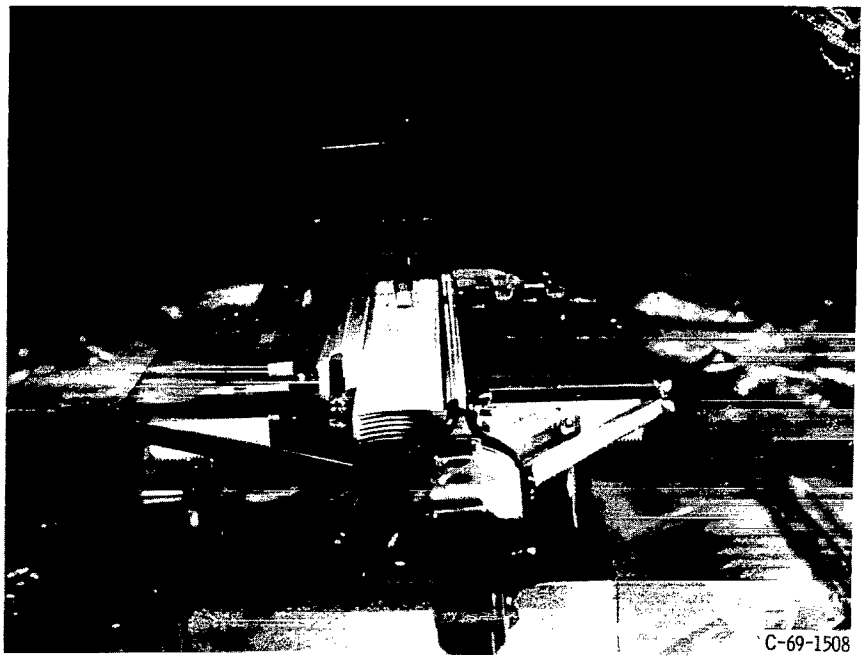


Figure 10. - Fuel-element disassembly fixture.

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